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REVIEW OF THE NRU SEISMIC UPGRADE PROGRAM

PREPARED FOR:

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REVIEW OF NRU SEISMIC UPGRADE PROGRAM

1.0 INTRODUCTION

The Atomic Energy Control Board, AECB has asked J.D. Stevenson, Consulting Engineer to perform a review of the status and ground motion and acceptance criteria being used for the Seismic re-evaluation of the NRU Reactor Facility (as shown in Figure 1) at the Chalk River Laboratory site. This review is intended to address the following areas:

Seismic

1. Ground motion defined for the re-evaluation of safety related structures, systems and components at the NRU facility (ABE seismic input).
2. Seismic ground motion defined for the design of seismic upgrades at the NRU facility (DBE seismic input).
3. The effect of soil-structure interaction foundation conditions at the site.
4. Review of structural evaluation and design criteria established by AECL for resolution of any seismic risk issues, seismic evaluation of existing structures and design of any new structures, systems or components at the NRU site.

The specific criteria used in the reevaluation and upgrading of the NRU facility are contained in Section 3.0.

It is recommended that a design review be performed to determine that the seismic input and evaluation and design criteria developed by AECL and reviewed herein, has been applied in an acceptable manner to the evaluation and design sample of at least one structure, one mechanical and one electrical component at the NRU facility. The detailed design review of individual Structures, Systems, and Components is beyond the scope of this project.

2.0 GROUND MOTION AT THE NRU CHALK RIVER SITE

2.1 Seismic Ground Motion on Rock

A great deal of effort in recent years has been expended to probabilistically define ground motion at North American sites on a generic basis. The latest such effort that I am aware of is the NEHRP-1997 Spectral Response Contours for the U.S. for 5.0 percent damping at 5.0 Hz (0.2 sec) for a 10 percent and 2 percent probability of exceedence in 50 years. This translates to 0.0021/yr or 0.0004/yr median probability of exceedence respectively. This median probability of exceedence also defines a median return period of 475 years and 2500 years respectively.

The 1995 edition of National Building Code of Canada, NBCC[1] uses the 10 percent probability of exceedence in 50 years requirement for ordinary conventional structures. Therefore, the median probability of exceedence in accordance with NBCC requirements defines a 0.0021/yr probability of exceedence or 475 year return period for ordinary Canadian structures.

It is noted that a 0.001/yr probability of occurrence has been established at the NRU site for a Design Basis Earthquake, DBE (0.24g ZPGA[1]). A second Assessment Basis Earthquake (ABE) has been established for the evaluation of existing structures, systems and components. The return period for the ABE is approximately the same as that for the DBE except that inelastic demand reduction factors as defined in the NBCC are applied to modify the ABE spectral shape.

Please note I have not evaluated the correctness of the 0.07g ZPGA for the ABE or 0.24g ZPGA as specified in the SAR Section 2.4[(2)]. I have reviewed Ref. 3 which is referenced in the NRU SAR. However, Ref. 3 does not provide any basis for the g values selected. Ref. 3 makes reference to AECL documents TR-314 and AECL-9251 which have not been provided for review which might show how the 0.07g ZPGA for ABE and 0.24g ZPGA values were obtained.

There also have been independent probabilistic seismic ground motion estimates developed for rock at the Chalk River Site by Dr. G. Atkinson.[(4)] It is also understood from AECL comments that a more recent study by Dr. Atkinson confirms the 0.24g ZPGA for the 0.001 annual probability of exceedence levels. Based on my understanding of the seismology of ENA, I believe the ZPGA values selected for the ABE and DBE seismic events are reasonable.

In my opinion an existing nuclear facility such as NRU reactor facility should be evaluated for a seismic input which is somewhat more stringent than the NBCC for ordinary structures. In Figure 2 is shown a comparison between the ABE and DBE spectra. The modified DBE spectra considers the following effects:

- Use of median rather than median plus one standard deviation shaped spectra[(5)]
- An inelastic energy reduction factor (Fu) equal to 3.0 as determined from Table 4.1.9.B Force Modification Factors of NBCC-95.

3.0 REVIEW OF ACCEPTANCE CRITERIA USED FOR DESIGN AND ANALYSIS

The basic acceptance criteria for the NRU Seismic Re-evaluation are contained in two criteria documents:

- Jategaonkar, R. And Legris, L., "Seismic Qualification Status Report," A-14157-TN-20 AECL, Feb. 1997.

- Jategaonkar, R., and Shah, P.J., "Design for Earthquakes (Seismic Qualification at CRL)," AECL, March 1993.

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[1]ZPGA = Zero Period Ground Acceleration.

3.1 APPLICABLE DOCUMENTS

The latest issues of the following documents form part of the Reference 6 seismic qualification procedure to the extent specified herein.

A-14115-TN-1,	Design Basis Earthquake Compatible Time-History for CRL Site (Engineering and Design Division Document)
CAN/CSA-N285.0,	General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants
CSA-B51,	Boiler, Pressure Vessel, and Pressure Piping Code
CSA CAN3-N287.3,	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CSA CAN3-N289.1,	General Requirements for Seismic Qualification of CANDU Nuclear Power Plants
CSA CAN3-N289.2,	Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants
CSA CAN3-N289.3,	Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants.
CSA CAN3-N289.4,	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants
NBCC 1995,	National Building Code of Canada

The following documents are referenced also in Ref. (6).

AECL-9251,	Chalk River Area Seismicity and its Implications for Low-Level Radioactive Waste Disposal Facility Siting
AECL TR-314,	Preliminary Analysis of Seismic Concerns for the Siting of a Shallow Land Burial Facility at CRNL
IAEA-TECDOC-348,	Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory (issued by the International Atomic Energy Agency, Vienna).

The NRU research reactor at Chalk River Laboratories was designed in the mid 1950's to construction standards in place at that time. No provisions were made in the original design for earthquake loading.

Modern CSA-CAN3-289[7] standards for design of new nuclear power plants include specific requirements for considering the effects of earthquakes. Provisions are also given in current Canadian building codes for the design of seismic-resistant structures, systems and components.

The present seismic assessment of the NRU facility is performed in order to determine the current "as is" seismic strength capacities of building structures and safety-related systems. In addition, the assessment identifies vulnerable components which may suffer earthquake-induced damage, and provides guidance for strengthening and upgrading the same components.

The scope of seismic work covers the following main phases of assessment or upgrade.

a) Seismic assessment of the reactor structures and selected safety-related systems is the Design Basis Earthquake, DBE. This phase of work is the original seismic assessment of NRU.

b) Seismic assessment of the reactor and reactor concrete structure is to a less severe seismic event, called the Assessment Basis Earthquake, ABE which permits the use of inelastic demand reduction factors as defined in the NBCC. This work forms the basis for the need to develop various NRU seismic upgrades.

c) seismic qualification work performed to date for the NRU upgrades:

   (i) Qualified Emergency Water Supply System
   (ii) Second Reactor Trip System and
   (iii) Qualified Emergency Response Centre

d) Seismic assessment of other components including the fuel storage bays, the fuel rod flask and the J-rod flask, the main building crane and the reactor bridge.

Since the construction of NRU in the 1950's, seismic design requirements for nuclear facilities in Canada have evolved considerably. Current requirements for seismic design of nuclear power plants are documented in the CSA standards listed in this subsection.

The general requirements set forth for seismic design safety of a CANDU Nuclear Power Plant are as follows:

a) safety-related structures and systems shall be seismically qualified;

b) failure or dislocation of other structures and systems not required for nuclear safety shall not affect the operation of any safety-related system;

c) earthquake effects shall not lead to unacceptable radiation exposure to operators or the public.

To achieve these requirements for a CANDU Nuclear Plant, the following apply:

a) it shall be possible to safely shut down the reactor and maintain it in that state indefinitely.

b) decay heat shall be capable of being removed from the fuel during this shutdown period;

c) the containment structure shall remain intact so as to limit radioactivity release.

In general these same general requirements were applied to the evaluation of NRU reactor facilities in the development of NRU seismic qualification groupings.

3.2 SEISMIC QUALIFICATION GROUPS

Process systems, equipment and structures at CRL including the NRU reactor are divided into the following three groups for seismic design considerations. These groups have been developed by use of classifications given in CAN/CSA-N285.0[8] so that confirmation of design adequacy could be easily verified.

The appropriate CRL-NRU seismic design group for structures, systems and components were established in consultation with the NRU operating groups and the AECL Research Regulatory and Quality Assurance Services (RQAS) Division.

3.2.1 GROUP A

Group A includes all safety related process systems, containment systems, equipment and structures classified as Nuclear Class 1,2,3 and 4 in CAN/CSA-N285.0, Clause 5.

3.2.2 GROUP B

Group B includes all process systems, equipment and structures with limited radioactive inventory (such as described in IAEA document TECDOC-348[9]). It also includes all systems and structures whose failure or collapse would directly or indirectly result in the failure of Group A systems and/or structures, and result in limited release of radioactivity exceeding the dose limits specified by the AECB.

An equivalent to Group B classification internationally is often used for Structures, Systems and Components, SSC associated with storage of high level radioactive waste. It is also used to provide service, secondary or backup systems for safety class Group A systems. An example of the standards used in the construction of Group B SSC is defined for high level waste in the U.S.

from R.G. 1.143[10] as shown in Table 4. It should be noted in Table 4 that storage vessels associated with high level waste are defined as ASME B&PVC Section VIII, Division 1 rather than meet the more stringent ASME B&PVC Section VIII, Division 2 requirements.

3.2.3 GROUP C

Group C includes all process systems, equipment and structures classified as Class 6 systems in CAN/CSA-N285.0, Clause 5. It also includes systems and structures not included in CRL Group A or B.

The NRU reactor building and reactor structure are classified as CRL Group A. The seismic assessment or components of these structures is performed in accordance with CSA N289.3[11]. Other safety-related systems are classified as CRL Group B and the seismic assessment of these systems is made on the basis of ASME Boiler and Pressure Vessel Code Section III Class 2 or Section VIII-Div. 2 requirements.

In my opinion the requirement to meet ASME BPVC Section III, Class 2 or Section VIII, Division 2 requirements for existing systems and components is inconsistent with the seismic Group B definition. I believe the more appropriate ASME BPVC classification would be the less rigorous Section III, Class 3 or Section VIII, Division 1.

3.3 METHODS OF SEISMIC QUALIFICATION

Seismic qualification of SSC at NRU means the verification of seismic design adequacy, by testing and/or analysis, to demonstrate the SSC's ability to perform its intended function during and/or following the designated earthquake[6]. For qualification by analysis, structural response is initially determined using the response spectrum technique, followed by a strength evaluation in accordance with the appropriate material code; for concrete structures CAN3-A23.3[6], and for steel structures CAN/CSA-S16.1.

In my opinion qualification of existing systems and components may use analysis or test results. However, it is generally more effective in existing facilities to use comparisons to similar systems and components which have experienced seismic demands equal to or greater than the designated earthquake without damage and less of safety function.[6]

3.4 SEISMIC QUALIFICATION GROUND MOTION

All new structures and equipment, which have a requirement for seismic qualification, are being qualified to the DBE assuming a ductility of 1.0 (elastic behavior). Existing structures, systems and components that support new equipment required safety functions, however, are being seismically qualified to the ABE which assumes a ductility factor of 3.0 consistent with the National Building Code of Canada.[1]

This commitment is somewhat more conservative when compared to current international practices used in evaluation existing nuclear power stations for revised or updated seismic requirements. In Table 5 is given an example of the current IAEA[11] guidance for evaluation of existing Russian designed NPP SSC. The difference between ductilities defined by the IAEA and those used by the NBCC are a result of the somewhat higher standards of construction observed in Canadian plants as compared to plants in Eastern Europe.

4.0 WALKBY EVALUATION

I performed a walkby (Seismic Safari) of the NRU reactor facility. The structure, systems and components of the facility appear to be well maintain and to be characteristic of good industrial detailing practices which have performed well in strong motion earthquakes. One observation of note is relative to seismic housekeeping practices. A number of laterally unsecured potentially rolling, sliding or overturning components were observed. It is strongly recommended that a procedure be developed and applied to prevent safety related SSC being impacted by other rolling, sliding or overturning equipment in the event of an earthquake.

5.0 CONCLUSIONS AND RECOMMENDATIONS

5.1 CONCLUSIONS

In general I have concluded that the NRU reactor evaluations and upgrade designs are proceeding in an acceptable manner, based on the criteria presented, when compared to other similar seismic reevaluations and upgrades of older nuclear facilities in North American, Western and Eastern Europe. However, I take exception to the reliance of analysis alone to qualify existing safety related systems and components. Such existing systems and components are more often evaluated by use of experience and test data for the seismic response of similar items. This is particularly true for items such as electrical equipment where required safety related operability can not be easily determined by analysis.

In addition, ductility greater than 1.0 with limited deformations into the inelastic range is often used in evaluation of existing nuclear facility SSC.[6,11,12]

An additional area of conservatism is the use of a median plus one standard deviation shaped ground response spectrum as defined in CSA-CAN3-289. This spectrum is meant for design of new safety related SSC at a nuclear power plant. For evaluation of existing SSC at nuclear facilities it is more common to use median amplification factors rather than median plus one standard deviation factors.

5.2 RECOMMENDATIONS

In my opinion, based on similar facility evaluations in North America and elsewhere, both the ABE and the DBE should be defined at the 0.001 probability of exceedence level. The DBE should be used for design of new safety related SSC in accordance with the elastic ground

response spectra defined in CSA-CAN3-289. However, the ABE used to evaluate existing SSC should consider limited inelasticity and median as opposed to median plus one standard deviation amplification factors in determining applicable spectral shapes in t he evaluation of existing SSC.

5.3 SUMMARY

(1) More attention needs to be made concerning seismic housekeeping on site. There are a number of instances where laterally unrestrained components capable of sliding, rolling, swinging or overturning in an earthquake are adjacent to safety related components. Adverse spatial interaction could result. A procedure for securing potentially moveable components should be prepared and implemented at the NRU facility.

(2) Some limited ductility should be permitted as is typically done for IAEA[11] and DOE[12] seismic reevaluation of existing nuclear facilities and in conformance with the NBCC.

(3) A median or best estimate rather than median plus one standard deviation should be defined as seismic spectral input for the reevaluation of existing facilities.[5]

(4) Making adjustments associated with the median to median plus one standard deviation and the application of Force Modification Factors, Fu to the elastic DBE spectral curve results in a modified spectral curve which is approximately equal to the current ABE spectral curve.

(5) Design reviews should be conducted on at least one NRU structure, mechanical and electrical component to assure design and or evaluation criteria are being properly applied as committed.

(6) Seismic reevaluation of systems and components seems to rely primarily on analysis. Seismic experience data in many instances is a better indicator of performance under seismic loads than analysis.

(7) Some of the industry standards quoted for evaluation or design were developed specifically for new design of large nuclear power plants, thus, appear to be excessively rigorous and costly which may create opposition to performing a particular upgrade.

6.0 REFERENCES

(1) National Building Code of Canada, 1995 Edition, National Research Council of Canada, 1996.

(2) AECL-MISC-300, "NRU Research Reactor Safety Analysis Report, "Vol. 1, Rev. 0, Atomic Energy of Canada Limited, 1998.

(3) Jategaonkar, R. and Shah, P.J., "Design for Earthquakes (Seismic Qualification) at CRL," AECL Engineering and Design Division, March 1993.

(4)[1] Atkinson, G.M., "Seismic Hazards at Chalk River Ontario," MX10-05600-010-TN, March 1989.

(5) Chen, J.T. et. al., "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407F Prepared for U.S. Nuclear Regulatory Research, June 1991.

(6) Jategaonkar, R. and Legris, L., "Seismic Qualification Status Report," A-14157-TN-20 AECL, Feb. 1997.

(7) CSA CAN 3-289, "Requirements for Seismic Qualifications of CANDU Nuclear Power Plants," Canadian Standards Association, Sept. 1981.

(8) CSA/N285.0, "General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants," Canadian National Standards Association.

(9) "Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory," IAEA Tech Doc 348, 1985.

(10) Regulatory Guide 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, Oct. 1979.

(11) IAEA/RU-869, "Technical Guidelines for Seismic Reevaluation Programme of the Armenian Nuclear Power Plant – unit 2," IAEA 1997.

(12) DOE Std. 1020, "Natural Phenomena Hazards Design and Evaluation Criteria for DOE Facilities," Rev. 1, U.S. Department of Energy, 1996.

TABLE 1.SEISMIC RISK CALCULATED FOR CRNL AREA

Location 45.90°N 77.45°W

Probability of exceedance per annum	0.01	0.002*	0.001
Peak horizontal acceleration (g)	0.072	0.167	0.241

TABLE 2: Reduction factors used to modify the DBE spectral shape and ZPGA values for use as a ABE spectra which depend on ductility and median rather than median plus one standard deviation amplification values.

Frequency	Inelastic Demand Reduction Factor[1] Fu	Spectral Shapes Median-Median Plus one Sigma
>10 Hz	3.0	1.2
5 Hz	3.0	1.3
2.5 Hz	3.0	1.3
<1.0 Hz	4.0	1.3

The reciprocal of these values multiplied together and then multiplied by the spectral ordinates of the DBE 5 percent damped response spectrum curve yields the modified DBE which could be used to define the ABE.

TABLE 3: DESIGN METHODOLOGY AND CODES AND STANDARDS FOR SEISMIC QUALIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

CRL GROUP	SEISMIC LEVEL	ANALYSIS METHOD	APPLICABLE CODE/STD FOR BUILDINGS & STRUCTURES	APPLICABLE CODE/STD FOR SYSTEMS & EQUIPMENT	REMARKS
A	DBE And ADE	Dynamic	CAN3-N289.3 CAN3-N287.3	CAN3-N289.3 CAN/CSA -N285 series	If testing is required for systems and equipment, use CAN3-N289.4
	or NBCC	See note (a) below	NBCC	NBCC	
B	DBE	Simplified Dynamic or Equivalent Static	IAEA-TECDOC-348 CSA A23.3 CSA S16.1	IAEA-TECDOC-348 Applicable CSA and ASME codes	See note (b) below
	Or NBCC	See note (a) below	NBCC	NBCC	
C	NBCC	See note (a) below	NBCC	NBCC or CSA B51	

ABBREVIATIONS:
DBE: Design Basis Earthquake
NBCC: National Building Code of Canada
SDE: Site Design Earthquake

NOTES:
(a) The NBCC specified a minimum base shear V (as described in Subsection 4.4 of procedure) and the lateral force distribution based on the weight distribution in the system as a simplified dynamic analysis.
(b) Where the applicability of Code IAEA-TECDOC-348 is in doubt for CRL Group B systems, use the codes and standards listed for CRL Group A.
(c) Early in the design phase, a proposed CRL Group in accordance with Subsection 4.2 of this procedure shall be established, and acceptance by the Safety Review Committee (SRC) shall be sought.
(d) The checklist questionnaire provided in Attachment B of this procedure shall be used to determine the proposed CRL Group.

TABLE 4: LIST OF STANDARDS FOR HIGH LEVEL RADIOACTIVE WASTE COMPONENTS AT U.S. NUCLEAR POWER PLANTS

EQUIPMENT	CODES			
	Design and Fabrication	Materials[1]	Welder Qualification and Procedures	Inspection and Testing
Pressure Vessels	AMSE Code Section VIII, Div. 1	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII, Div. 1
Atmospheric Tanks	ASME Code[3] Section III, Class 3, or API 650, or AWWA D-100[2]	ASME Code[2] Section II	ASME Code Section IX	ASME Code[3] Section III, Class 3, or API 650, or AWWA D-100[2]
0-15 PSIG Tanks	ASME Code[3] Section III, Class 3, or API 620[2]	ASME Code[2] Section II	ASME Code Section IX	ASME Code[3] Section III, Class 3, or API 620[2]
Heat Exchangers	ASME Code Section VIII, Div. 1 and TEMA	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII, Div. 1
Piping and Valves	ANSI B31.1	ASTM and ASME Code Section II	ASME Code Section IX	ANSI B31.1
Pumps	Manufacturers' Standards[4]	ASME Code Section II or Manufacturers' Standards	ASME Code Section IX (as required)	ASME Code[3] Section III, Class 3; or Hydraulic Institute

[1] Manufacturers' material certificates of compliance with material specifications may be provided in lieu of certified material.

[2] Fiberglass-reinforced plastic tanks may be used in accordance with appropriate articles of Section 10 of the ASME Boiler and Pressure Vessel Code for applications at ambient temperature.

[3] ASME Code stamp, material traceability, and the quality assurance criteria of Appendix B to 10 CFR Part 50 are not required. Therefore, these components are not classified as ASME Code Class 3.

[4] Manufacturers' standard for the intended service. Hydrotesting should be 1.5 times the design pressure.

TABLE 5 IAEA INELASTIC ENERGY ABSORPTION RATIO, F_μ[1] or DUCTILITY, μ

STRUCTURAL SYSTEM	F_μ or μ	
	A[2]	B[3]
MOMENT RESISTING FRAME SYSTEMS		
Concrete:		
Columns	1.25	1.50
Beams	1.5	2.00
Connections	1.0	1.50
Steel:		
Columns	1.0	1.0[4]
Beams	2.0	2.5
Connections	1.0	1.5
SHEAR WALLS		
Concrete	1.25	1.75
Reinforced Masonry	1.25	1.50
Unreinforced Masonry	1.0	1.00
BRACED FRAMES		
Concrete:		
Columns	1.25	1.50
Beams	1.5	2.00
Diagonal Bracing		
Tension	1.5	3.0
Compression	1.0	1.0
Connections	1.0	1.5
Steel:		
Columns	1.0	1.0
Beams	2.0	2.5
Connections	1.0	1.5
Adequately Anchored Passive Electrical and Mechanical Equipment:		
Bent plate panels	1.5	1.5
Steel angles framing	2.0	2.0
Steel housings	2.0	2.0
Cast Iron	1.0	--
Piping, Conduit, Instrument Tubing and HVAC Duct[2]:		
Butt joined grove welded steel pipe	1.5	2.0
Socket welded pipe	1.25	1.5
Threaded pipe	1.25	1.25
Conduit	1.50	1.25
Instrument Tubing	1.50	1.50
Cable trays,	1.50	1.50
HVAC duct	1.25	1.50
Distribution System Supports		

[1]Th relationship between the inelastic energy absorption ratios, F_μ and ductility μ is given in Ref. 9 as a function of frequency.

[2]These Inelastic Energy Absorption ratios are recommended for use with existing VVER Structures, Systems, and Components.

[3]These Inelastic Demand-Capacity ratios are recommended for use with structures designed using joint ductile requirements contained in ACI-318-92 Chapter 21 for concrete or the SEAOC Code for structural steel or their national equivalent.

[3]For components in axial compression with K1/r rations less than 40, R_μ may be taken as 1.25.

[4]For metal pressure retaining components if stresses are limited to ASME B&PVC III – 1992 or earlier code allowables otherwise Fu and μ=1.0.

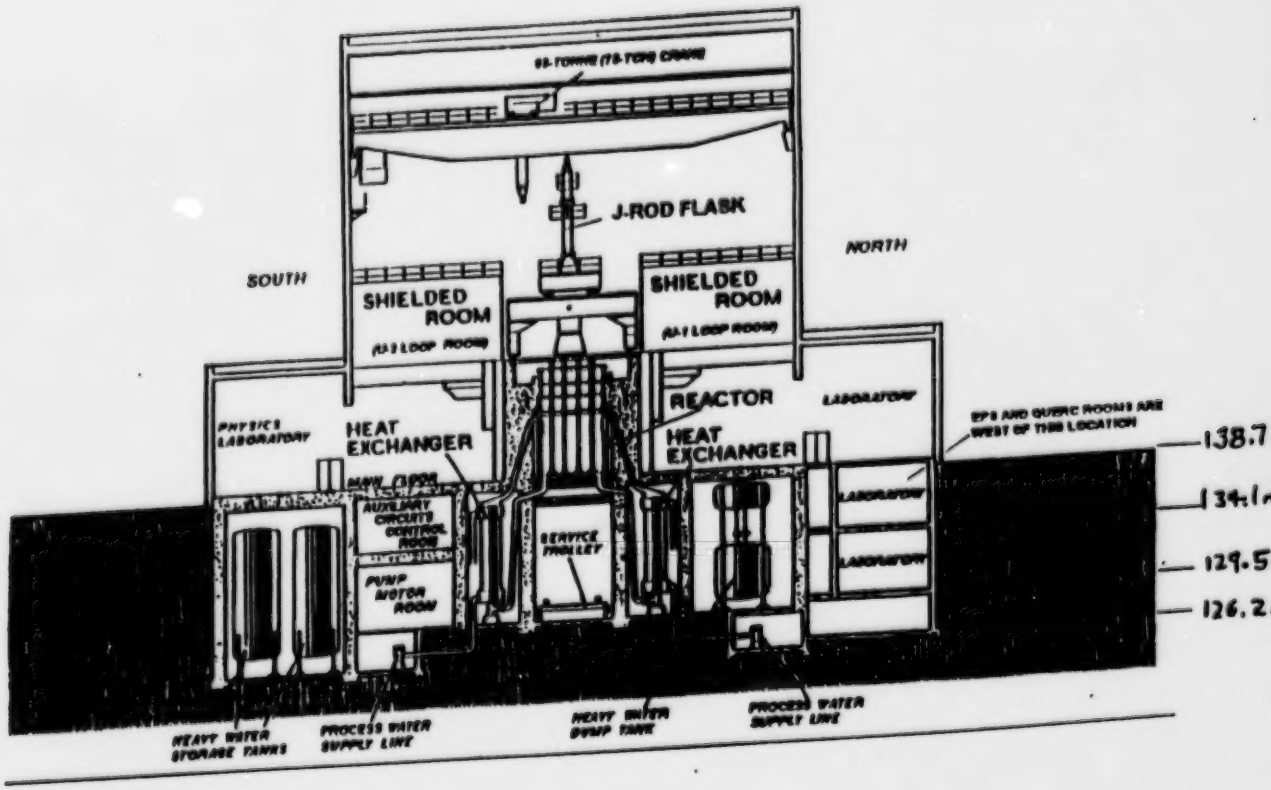

FIGURE 1 : NRU Reactor Building: South-North cross-section.

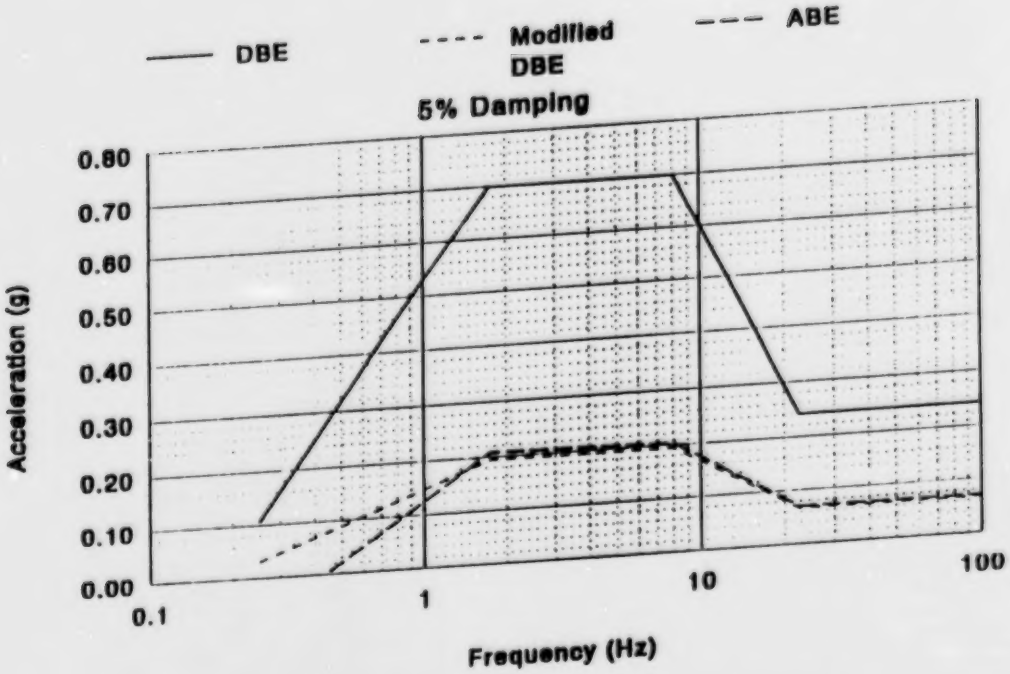

Figure 2 – Comparison of DBE, ABE, and Modified DBE for the NRU Seismic Upgrade